APPENDIX C

SPENT NUCLEAR FUEL BACKGROUND AND INVENTORY

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APPENDIX C. SPENT NUCLEAR FUEL BACKGROUND AND INVENTORY

C.1 Background

C.1.1 GENERAL CHARACTERISTICS

In nuclear reactors a process occurs known as the fission chain reaction, in which atomic nuclei in reactor fuel respond to collisions with neutrons by splitting into two major fragments and two or three neutrons. The neutrons can interact with other fuel nuclei, thereby continuing the chain reaction.

In comparison to a chemical reaction involving the same mass, a nuclear reaction releases a large amount of energy, mostly the kinetic energy of the fission fragments and neutrons and the subsequent radioactive decay of the fission fragments (fission products). This energy makes nuclear fission an attractive source of energy for commercial power producers. DOE operated its production reactors principally because the neutrons caused nonfission nuclear reactions of interest to national defense (i.e., isotopic transmutation). Research reactors use the fission process to produce medical isotopes or for other research purposes.

Nuclear fuel must contain atoms that can be fissioned (called fission atoms). Fission atoms are fissioned by low-energy (thermal) neutrons. Therefore, to maintain the chain reaction, the high energy, fast neutrons produced by fission must be slowed to low-energy, thermal neutrons. The process for slowing down the neutrons is called moderation: water, graphite, and heavy water are used as moderators.

Uranium-235 is the fissile atom used most often for nuclear fuel; however, other fissile materials (uranium-233, plutonium-239, and plutonium-241) can be used in nuclear reactors. Uranium-235 represents only about 0.7 percent of the atoms of natural uranium, which is primarily uranium-238. Therefore, many reactors use fuel that has an enriched uranium-235 content. Commercial power reactors typically use fuels enriched to approximately 2 to 4 percent. Noncommercial reactors, depending on their purpose, use fuel enriched to as much as 93 percent uranium-235. Low-enriched uranium (LEU) has an enrichment below 20 percent; highly enriched uranium (HEU) is enriched 20 percent or higher. The fuels discussed in this EIS are primarily highly enriched uranium fuels.

The uranium in nuclear fuels generally is clad with a metal to protect it from chemical reactions with the moderator water and to prevent the release of fission products to the water. Zirconium, stainless steel, and aluminum are common cladding materials. Most of the SNF analyzed in this EIS (about 48 metric tons heavy metal [MTHM]) is aluminum-clad; the remainder is clad with stainless steel or zirconium.

Inside the cladding, the fuel is often in the form of a ceramic, an alloy that combines uranium with aluminum, metallic uranium, or a uranium oxide or silicide. The fuel can be assembled as parallel plates, concentric tubes, bundles of rods or pins, or other designs. Each assembly has mounting and lifting hardware, structures to direct coolant and moderator flow, and in some cases the capability to install neutron absorbing material and instrumentation. Usually a number of fuel assemblies make up a complete reactor core.

Spent nuclear fuel (SNF) is fuel that has been irradiated in a reactor and contains fissile atoms and fission products. SNF management must consider four fuel characteristics: radiation fields, heat generation, criticality, and chemical stability (corrosion resistance). As the fuel is irradiated in a reactor, much of the uranium is burned, resulting in the production of fission products. These fission products are radioactive; that is, they do not undergo fission but they radiate energy and transmute to other elements. SNF has very high radiation fields, especially for a

EC period of time immediately after it is removed from the reactor. After a period of decay, as the short-lived fission products decay away, the radiation fields decrease, but the fuel is still highly radioactive and requires management for many years.

The heat from the radioactive decay of fission products (decay heat) can produce very high temperatures, requiring fuel recently removed from a reactor to be placed in underwater storage for cooling. Without active cooling, the fuel could overheat and melt or damage the cladding. After a sufficient cooling time that depends on the burnup of the fuel and its composition, fuel assemblies can be stored dry. Dry fuel storage technologies must be designed to release residual decay heat.

Long-term storage of SNF in water can lead to corrosion of the fuel cladding. Careful control of water chemistry can reduce the rate of corrosion. Aluminum-clad fuels, which are considered in this EIS, are more prone to corrosion in water than are stainless-steel or zirconium-clad fuels.

Most SNF could undergo a fission chain reaction. However, the fuel density, geometry, temperature, and moderation must support fission, or the chain reaction would not occur because too many neutrons would be absorbed or otherwise lost.

When a reactor is producing enough neutrons to support a chain reaction, it is termed "critical." Criticality occurs when fissile material begins to undergo a chain reaction. SNF management must consider the potential of the fuel to create an unwanted criticality.

SNF can be chemically processed to recover transmitted isotopes for defense or commercial purposes and the fissile and fertile material for conversion into more nuclear fuel.

C.1.2 RECENT SPENT FUEL MANAGEMENT ACTIONS

In 1992, DOE decided to phase out defense-related SNF processing. Subsequently, the Department began to establish programs to manage DOE SNF that were no longer based on the production of strategic nuclear material. DOE identified the initial components of this plan in the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995a) (hereafter referred to as the Programmatic SNF EIS). The Record of Decision for this environmental impact statement (EIS) (60 FR 28680) stated in part that DOE would consolidate the management of its aluminum-clad SNF at the Savannah River Site (SRS) and would consolidate nonaluminum-clad fuels at the Idaho National Engineering and Environmental Laboratory. As a result, about 20 MTHM of stainless-steel and zirconium-clad SNF stored at SRS was designated for shipment to the Idaho National Engineering and Environmental Laboratory for management. In addition, DOE decided to ship about 10 MTHM of aluminum-clad SNF to SRS from domestic, and DOE research reactors, and the Idaho National Engineering and Environmental Laboratory.

However, in the Programmatic SNF EIS Record of Decision DOE made no decisions on the technologies it would apply to the management of SNF at the designated storage sites. The Record of Decision stated that the selection of SNF stabilization technologies and the preparation of SNF for ultimate disposition would be the subject of site-specific and fuel-type-specific evaluations prepared in accordance with the National Environmental Policy Act and tiered from the Programmatic SNF EIS (DOE 1995a).

In October 1995, DOE assessed the environmental impacts of stabilizing certain nuclear materials at SRS that presented potential environment, safety, and health vulnerabilities (DOE 1995b). The material evaluated by DOE included SRS production reactor SNF stored in the reactor disassembly basins and research reactor SNF stored in the Receiving Basin for Offsite Fuel. The Department decided to stabilize SNF that presented potential environmental, safety, and health vulnerabilities by processing the material through the existing chemical separations facilities at SRS. Under these decisions (60 FR 65300, 61 FR 6633, and 62 FR 17790), about 175 MTHM of the approximately 195 MTHM of SNF at SRS will be stabilized. After stabilization, the resulting material will be treated and managed so that it is acceptable for permanent disposition once those decisions are made. DOE concluded the remaining material, all of which was stored in the Receiving Basin for Offsite Fuel, was stable and could remain as is for several years pending disposition decisions. In addition, DOE decided some of the stable material might have programmatic value, that is, be of use to future DOE missions. Mark-18 targets stored in the Receiving Basin for Offsite Fuel could be shipped to other DOE sites for programmatic uses, including irradiation for transuranium isotope production (primarily for National Aeronautics and Space Administration use) and defense stockpile stewardship activities.

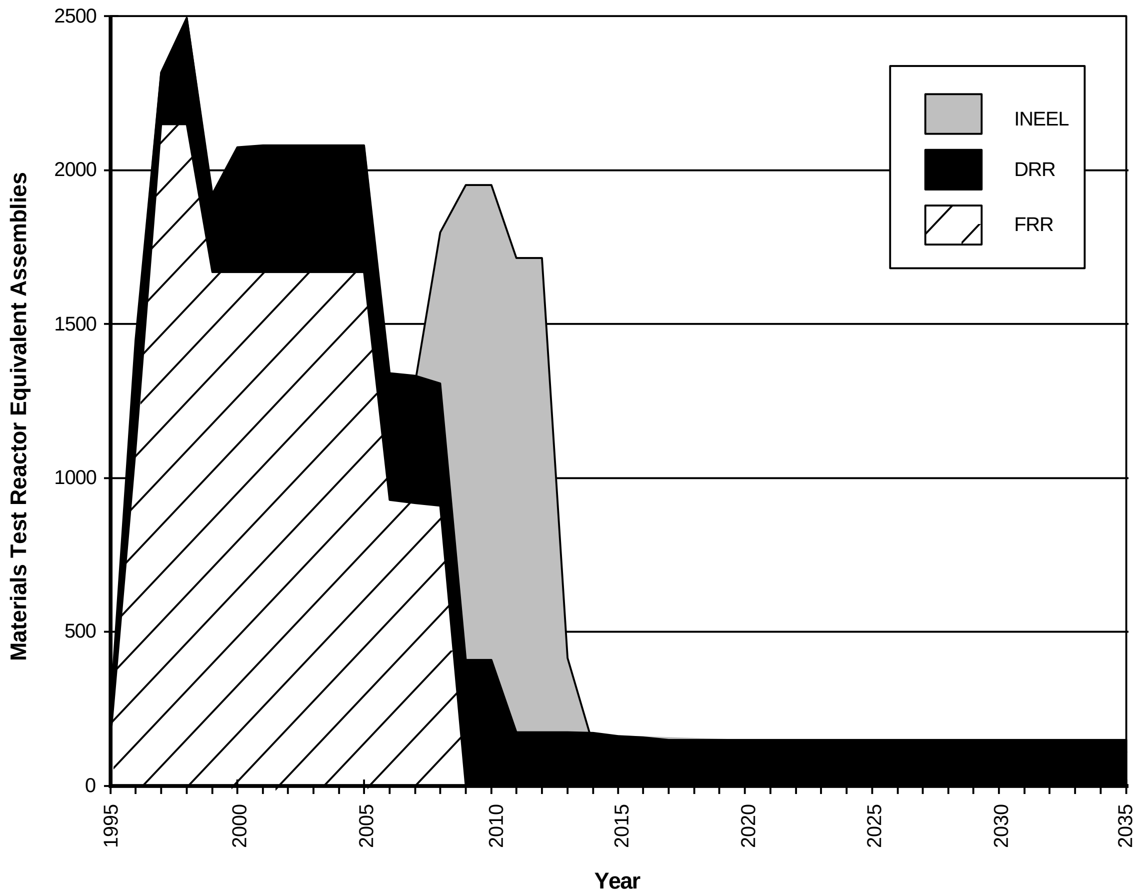
In May 1996 DOE issued its Record of Decision (61 FR 25092) for the *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor [FRR] Spent Nuclear Fuel* (DOE 1996a) (hereafter referred to as the FRR EIS). The Department decided to accept and manage foreign research reactor SNF that contains uranium enriched in the United States. In keeping with its 1995 programmatic decision (60 FR 28680), DOE decided it would manage the aluminum-clad portion of the foreign research reactor SNF, about 18 MTHM, at SRS. Under the foreign research reactor receipt program, shipments from foreign reactors to SRS began in September 1996 and could continue for as long as 13 years. At present, SRS is receiving this fuel in either the Receiving Basin for Offsite Fuel or the L-Reactor Disassembly Basin. Figure C-1 shows projected receipts of aluminum-clad SNF at SRS from foreign and domestic sources, based on 1996 estimates. Because some countries may choose not to participate in the return of foreign SNF, the amount of aluminum-based foreign SNF to be managed at SRS may be less.

The May 1996 decision to accept foreign research reactor SNF for management in the United States (61 FR 25092) stated that DOE would issue a separate Record of Decision, after appropriate environmental reviews, to announce its plans for the management of such fuel. The Department committed to the aggressive pursuit of one or more new packaging or non-processing technologies that would put foreign research reactor SNF in a form or container suitable for disposal in a geologic repository. DOE also committed to place foreign research reactor SNF in dry storage at SRS (after required treatment or packaging) pending offsite storage or disposal. DOE also stated that if a new treatment technology was not ready for implementation by 2000, DOE would consider the chemical separation of some foreign reactor SNF that would blend the material down to low-enriched uranium in F Canyon at SRS. DOE might then place it under International Atomic Energy Agency safeguards.

C.2 Inventory

C.2.1 PHYSICAL INVENTORY

As Figure C-1 indicates, most SNF receipts would occur before 2015; however, SRS will continue to receive small amounts over the entire period of analysis (until 2035). There is great variety in the SNF that SRS must manage over the next 40 years. Therefore, DOE has categorized the SNF into six groups to facilitate analysis. The SNF in each category should receive nearly identical management. The basis for the categorization was often the size of the fuel in relation to packaging; however, other considerations were included such as physical characteristics, chemical characteristics, and radionuclide content. For example, one category includes all SNF in powder form. The following subsections describe the six fuel groups and list the SNF inventory associated associated with each group. Receipts per year will be aproximately 150 Materials Test M4-13



INEEL = Idaho National Engineering and Environmental Laboratory
DRR = Domestic Research Reactor
FRR = Foreign Research Reactor
Source: DOE 1996b.

Figure C-1. Projected receipts of SNF at the Savannah River Site.

Reactor-like Elements from domestic reactors and 12 High Flux Isotope Reactor assemblies from Oak Ridge.

C.2.1.1 <u>Group A: Uranium and Thorium Metal Fuels</u>

Group A consists mostly of chemically reactive uranium and thorium metal fuels. Many of the fuel elements are declad, and much of this group consists of depleted or natural uranium. As indicated in Table C-1, Group A fuels consist of four fuel types. The Experimental Breeder Reactor-II Blanket Fuels have been declad and the depleted uranium slugs placed in aluminum cans. The Advanced Reactivity Measurement Facility (ARMF) Core Filter Block is a 6 × 6 × 24-inch (15.2 × 15.2 × 61-centimeter) block of depleted uranium. The Sodium Reactor Experiment fuel consists of declad thorium metal placed in 3.5-inch (8.9-centimeter) diameter by 110-inch (279-centimeter) long cans. The Mark-42 targets are unirradiated tubes of plutonium oxide in an aluminum matrix approximately 3.7 inches (9.4 centimeters) in diameter and 168 inches (426 centimeters) long.

C.2.1.2 <u>Group B: Materials Test Reactor-Like Fuels</u>

Group B is comprised mostly of Materials Test Reactor fuels, as described in Section 1.5 and

Figure 1-3, plus a few other fuels of similar size and composition. Table C-2 lists the Group B inventory

Table C-1. Inventory of Group A SNF.

Name	Items	Units	Location
Experimental Breeder Reactor Blankets	59	Cans	SRS Wet Basins[a]
ARMF Core Filter Block	1	Filter	INEEL
Sodium Reactor Experiment	36	Cans	SRS Wet Basins
Mark-42	16	Bundles	SRS Wet Basins

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.
ARMF = Advanced Reactivity Measurement Facility
INEEL = Idaho National Engineering and Environmental Laboratory

C.2.1.3 Group C: HEU/LEU Oxides and Silicides Requiring Resizing or Special Packaging

Group C fuels are similar in composition to Group B fuels in that they are aluminum-clad, highly enriched uranium and low enriched uranium oxides and silicides, but their size or shape precludes packaging without resizing or special packaging considerations. Some of the Group C SNF is smaller in diameter and longer than Group B fuels. Other fuel in this group is larger than Group B fuels in both diameter and length and often comes in odd shapes such as 0.5-by-0.9-meter (1.5-by-3-foot) cylinders or spheres with a diameter of 74 centimeters (29 inches). Table C-3 lists Group C inventory.

C.2.1.4 Group D: Loose Uranium Oxide in Cans

Group D fuels consist of loose uranium oxide and fission products in aluminum cans. Table C-4 lists the Group D inventory.

The Sterling Forest Oxide material in this fuel group is a residue of highly enriched uranium, plutonium, fission products, mixed oxides (chromium, nickel, iron, barium), barium acetate, and barium nitrate which resulted from the production of medical isotopes (primarily molybdenum 99). The material was plated on the inside of stainless steel tubes when it was irradiated. The material was then removed from the tubes with an acid flush and the uranium was recovered from a nitrate-sulfate solution, after eliminating the sulfate by precipitating with barium acetate and filtering. The filtrate was evaporated and pyrolyzed at 300ºC to an oxide form in an aluminum can. The can was sealed and shipped to SRS where it was placed into storage in the Receiving Basin for Offsite Fuels. Both the can and the oxide powder it contains are readily dissolved in acid.

The other items in this fuel group are liquid targets that DOE assumes would be converted to oxide prior to shipment to SRS.

C.2.1.5 Group E: Higher Actinide Targets

Group E contains target materials used to generate radionuclides with atomic numbers beyond that of uranium. The targets are placed in nuclear reactors and irradiated with neutrons, which causes nonfission nuclear reactions. These targets are aluminum-clad plutonium oxide that contain significant quantities of americium and curium, which react under neutron irradiation to produce even higher atomic number elements such as californium. Table C-5 lists the Group E inventory.

C.2.1.6 Group F: Non-aluminum Clad Fuels

Group F comprises the large variety of non-aluminum-clad SNF at SRS that DOE must ship to the Idaho National Engineering and Environmental Laboratory under the Record of Decision for the Programmatic SNF EIS (DOE 1995a). Table C-6 lists the Group F inventory.

C.2.2 RADIONUCLIDE INVENTORY

The six SNF groups that DOE would manage at SRS possess diverse chemical, physical, and radiological characteristics. There is also diversity within any single fuel group. In the absence of detailed radionuclide characterization of the fuel, DOE has simplified the analyses for this EIS by developing an analytical construction called a Reference Fuel Assembly (Garrett et al. 1995). The Reference Fuel Assembly is used as a standard reference for scaling fuel group characteristics. This assembly is a composite of depleted uranium, highly enriched uranium, and special target radionuclides.

To determine the radionuclide inventories of each fuel group, DOE calculated the ratio of radioactivity of each nuclide in the Reference Fuel Assembly to the fissile mass of the Reference Fuel Assembly; multiplied the resulting ratios by the fissile mass of the fuel groups to obtain nuclide-specific inventories for each fuel group. DOE performed an identical calculation based on total heavy metal mass rather than fissile mass of each fuel group. DOE conservatively used the calculation (fissile mass ratio or total heavy metal mass ratio) that yielded the largest value of each radionuclide to calculate the inventory of each radionuclide for the fuel group. Scaling by fissile mass is important because the fission products potentially produce most of the radiological impacts. Scaling by heavy metal mass is important because heavy metal mass is an indicator of processing time and provides appropriate representation of Group A fuels which contain little fissile mass. Table C-7 lists the results of these calculations.

Table C-2. Inventory of Group B SNF.

Name	Items	Units	Location
FRR MTR	10,812	Assemblies	Argentina, Australia, Austria, Brazil, Canada, Chile, Colombia, France, Germany, Greece, Indonesia, Iran, Israel, Italy, Japan, Netherlands, Pakistan, Peru, Philippines, Portugal, South Africa, Spain, Sweden, Switzerland, Taiwan, Thailand, Turkey, United Kingdom, Uruguay, Venezuela
DRR MTR	11,799	Assemblies	LANL, INEEL, ORNL, Brookhaven National Laboratory, Massachusetts Institute of Technology, Georgia Institute of Technology, Iowa State University, University of Massachusetts-Lowell, University of Michigan, Missouri University, Purdue University, Rhode Island Nuclear Center, University of Virginia, Worcester, National Institute of Standards and Technology
MTR	Approximately 1,100	Assemblies	SRS Wet Basins[a]
Cylindrical MTR	145	Assemblies	Japan
Box MTR	28	Assemblies	Japan
Tube MTR	4,077	Assemblies	Australia, Belgium, Denmark, Germany
Missouri University Research Reactor	224	Assemblies	SRS Wet Basins, MURR, INEEL
Advanced Test Reactor	3,132	Assemblies	INEEL
Advanced Reactivity Measurement Facility	67	Assemblies	INEEL
University of Washington	26	Bundles	INEEL
Advanced Reactivity Measurement Facility Plate	15	Plates	INEEL
Sterling Forest Fuel	200	Assemblies	SRS Wet Basins

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.
b. This value changes with FRR and DRR ongoing receipts. Some double counting with FRR and DRR entries exists.

LANL = Los Alamos National Laboratory
ORNL = Oak Ridge National Laboratory
MTR = Materials Test Reactor
FRR = Foreign Research Reactor
MURR = Missouri University Research Reactor
DRR = Domestic Research Reactor
INEEL = Idaho National Engineering and Environmental Laboratory

Table C-3. Inventory of Group C SNF.

Name	Items	Units	Location
Mark-14	1	Can	SRS Wet Basins[a]
Oak Ridge Research Reactor	165	Assemblies	SRS Wet Basins
HWCTR	1	Can	SRS Wet Basins
Pin bundle	12	Bundles	Canada, Jamaica
Pin cluster	2,792	Clusters	Canada, South Korea
ZPTR	45	Assemblies	Cornell University
ZPR	17	Assemblies	Manhattan University
OSR	24	Assemblies	Ohio State
Argonaut	50	Assemblies	Florida
Reactor a-Haut Flux	90	Assemblies	SRS Wet Basins, France
High Flux Isotope Reactor	540	Assemblies	ORNL
High Flux Isotope Reactor	1	Can	SRS Wet Basins
TC BSR	32	Assemblies	ORNL/SRS
Tower Shielding Reactor	1	Element	ORNL
Tower Shielding Reactor	2	Cans	ORNL
TC Sandia Pulse Reactor	43	Assemblies	Sandia National Laboratories
Oak Ridge Reactor	9	Cans	ORNL/SRS

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.
BSR = Bulk Shielding Reactor
OSR = Ohio State Reactor
ZPR = Zero Power Reactor
ZPTR = Zero Power Test Reactor
ORNL = Oak Ridge National Laboratory
HWCTR = Heavy Water Components Test Reactor

Table C-4. Inventory of Group D SNF.

Name	Items	Units	Location
Sterling Forest Oxide	676	Cans	SRS Wet Basins[a]
Other non-MTR targets	6,750	Cans	Canada, Belgium, Argentina, Indonesia

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.
MTR = Materials Test Reactor

Table C-5. Inventory of Group E SNF.

Name	Items	Units	Location
Mark-18	65	Assemblies	SRS Wet Basins[a]
Mark-51	60	Slugs	SRS Wet Basins
Other	114	Slugs	SRS Wet Basins

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

Table C-6. Inventory of Group F SNF.

Name	Items	Units	Current Location
Carolinas-Virginia Tube Reactor	3	Bundles	SRS Wet Basins[a]
Dresden	24	Sleeves	SRS Wet Basins
Dresden	6	Cans	SRS Wet Basins
Elk River Reactor	38	Bundles	SRS Wet Basins
LWR Samples	5	Cans	SRS Wet Basins
H. B. Robinson	1	Can	SRS Wet Basins
Saxton	13	Bundles	SRS Wet Basins
Saxton	3	Cans	SRS Wet Basins
Saxton	3	Test Tubes	SRS Wet Basins
Vallecitos	2	Bundles	SRS Wet Basins
Babcock & Wilcox Scrap	1	Can	SRS Wet Basins
EBR-II (ANL-MXOX)	1	Cans	SRS Wet Basins
EBWR	6	Cans	SRS Wet Basins
EBWR	4	Bundles	SRS Wet Basins
EBWR	288	Assemblies	SRS Wet Basins
EPRI	1	Can	SRS Wet Basins
GCRE	6	Cans	SRS Wet Basins
GCRE	66	Assemblies	SRS Wet Basins
HWCTR	34	Slugs	SRS Wet Basins
HWCTR	87	Cans	SRS Wet Basins
HWCTR	57	Assemblies	SRS Wet Basins
HWCTR	22	Bundles	SRS Wet Basins
HWCTR	9	Test Tubes	SRS Wet Basins
HTRE	13	Cans	SRS Wet Basins
ML-1	68	Assemblies	SRS Wet Basins
ORNL S1W-1 rods	3	Cans	SRS Wet Basins
ORNL Mixed Oxide (BW-1)	1	Can	SRS Wet Basins
Shippingport	127	Pins	SRS Wet Basins
SPERT-3	3	Cans	SRS Wet Basins
Sodium Reactor Experiment (Carbide)	1	Can	SRS Wet Basins
CANDU	3	Cans	SRS Wet Basins
CANDU	56	Rods	SRS Wet Basins

a. Receiving Basins for Offsite Fuel or L-Reactor Disassembly Basin.

EBR = Experimental Breeder Reactor
HWCTR = Heavy Water Components Test Reactor
CANDU = Canadian Deuterium-Uranium Reactor
LWR = Light Water Reactor
EBWR = Experimental Boiling Water Reactor
ANL-MXOX = Argonne National Laboratory Mixed Oxide
ORNL = Oak Ridge National Laboratory
GCRE = Gas Cooled Reactor Experiment
HTRE = High Temperature Reactor Experiment
ML-1 = Mobile Low Power Plant No. 1
SPERT-3 = Special Power Excursion Test-3
EPRI = Electric Power Research Institute

Table C-7. Radionuclide inventories based on the Reference Fuel Assembly (curies).

Nuclide[a]	Reference Fuel Assembly	Fuel Group A	B	C	D	E	F
H-3	51.6	2540	144,000	46,000	9,090	112	9,780
Kr-85	1,050	51,700	2,920,000	935,000	185,000	2,270	199,000
Sr-89	49.2	2420	137000	43800	8670	107	9320
Sr-90	8,080	398,000	22,500,000	7,200,000	1,420,000	17,500	1,530,000
Y-90	8,080	398,000	22,500,000	7,200,000	1,420,000	17,500	1,530,000
Y-91	213	10,500	593,000	190,000	37,500	461	40,400
Zr-95	454	22,400	1,260,000	404,000	80,000	983	86,000
Nb-95	1,010	49,800	2,810,000	899,000	178,000	2,190	191,000
Nb-95m	3.37	166	9,390	3,000	594	7.30	639
Tc-99	1.03	50.7	2,870	917	181	2.23	195
Rh-103m	1.96	96.6	5,460	1,750	345	4.25	371
Rh-106	21,100	1,040,000	58,800,000	18,800,000	3,720,000	45,700	4,000,000
Ru-103	2.17	107	6,040	1,930	382	4.70	411
Ru-106	21,100	1,040,000	58,800,000	18,800,000	3,720,000	45,700	4,000,000
Ag-110	2.32	114	6,460	2,070	409	5.03	440
Ag-110m	174	8,570	485,000	155,000	30,700	377	33,000
Cd-113m	6.95	342	19,400	6,190	1,220	15.1	1,320
Sn-119m	3.93	194	10,900	3,500	693	8.51	745
Sn-123	14.5	714	40,400	12,900	2,560	31.4	2,750
Sb-125	870	42,900	2,420,000	775,000	153,000	1,880	165,000
Te-125m	212	10,400	590,000	189,000	37,400	459	40,200
Te-127	34.7	1,710	96,600	30,900	6,110	75.2	6,570
Te-127m	35.4	1,740	98,600	31,500	6,240	76.7	6,710
Te-129	0.0012	591	3.34	1.07	0.211	0.0026	0.227
Te-129m	0.00185	911	5.15	1.65	0.326	0.00401	0.351
Cs-134	10,300	507,000	28,700,000	9,170,000	1,810,000	22,300	1,950,000
Cs-137	9,280	457,000	25,800,000	8,260,000	1,640,000	20,100	1,760,000
Ba-137m	8,780	433,000	24,500,000	7,820,000	1,550,000	19,000	1,660,000
Ce-141	0.0646	3.18	180	57.5	11.4	0.140	12.2
Ce-144	47,800	2,350,000	133,000,000	42,600,000	8,420,000	104,000	9,060,000
Pr-144	47,800	2,350,000	133,000,000	42,600,000	8,420,000	104,000	9,060,000
Pr-144m	574	28,300	1,600,000	511,000	101,000	1,240	109,000
Pm-147	18,800	926,000	52,400,000	16,700,000	3,310,000	40,700	3,560,000
Pm-148m	0.00893	0.44	24.9	7.95	1.57	0.0193	1.69
Sm-151	69.4	3,420	193,000	61,800	12,200	150	13,100
Eu-154	727	35,800	2,020,000	647,000	128,000	1,570	138,000
Eu-155	381	18,800	1,060,000	339,000	67,100	825	72,200
Tl-208	8.46	417	23,600	7,530	1,490	18.3	1,600
Pb-209	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Pb-211	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Pb-212	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Bi-211	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Bi-212	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Bi-213	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Po-212	15.1	744	42,100	13,400	2,660	32.7	2,860
Po-213	0.00855	0.421	23.8	7.61	1.51	0.0185	1.62

Table C-7. (continued).

Nuclide[a]	Reference Fuel Assembly	Fuel Group A	B	C	D	E	F
Po-215	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Po-216	23.6	1,160	65,700	21,000	4,160	51.1	4,470
At-217	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Rn-219	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Rn-220	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Fr-221	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ra-223	0.0166	0.818	46.2	14.8	2.93	0.036	3.15
Ra-224	23.6	1,160	65,700	21,000	4,160	51.1	4,470
Ra-225	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ac-225	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Ac-227	0.0171	0.842	47.6	15.2	3.01	0.037	3.24
Th-227	0.0164	0.808	45.7	14.6	2.89	0.0355	3.11
Th-228	23.5	1160	65,500	20,900	4,140	50.9	4,450
Th-229	0.00874	0.431	24.3	7.78	1.54	0.0189	1.66
Th-231	0.0114	0.562	31.8	10.2	2.01	0.0247	2.16
Th-232	0.0172	0.847	47.9	15.3	3.03	0.0373	3.26
Th-234	0.000216	0.0106	0.602	0.192	0.0381	0.000468	0.0409
Pa-231	0.228	11.2	635	203	40.2	0.494	43.2
Pa-233	0.0859	4.23	239	76.5	15.1	0.186	16.3
Pa-234m	0.000216	0.0106	0.602	0.192	0.0381	0.000468	0.0409
U-232	40.9	2,010	114,000	36,400	7,210	88.6	7,750
U-233	39.3	1,940	109,000	35,000	6,930	85.1	7,450
U-234	1.92	94.6	5,350	1,710	338	4.16	364
U-235	0.0114	0.562	31.8	10.2	2.01	0.0247	2.16
U-236	0.0329	1.62	91.6	29.3	5.80	0.0713	6.23
U-237	0.259	12.8	721	231	45.6	0.561	49.1
U-238	0.0842	4.15	235	75.0	14.8	0.182	16.0
Np-237	0.00881	0.434	24.5	7.85	1.55	0.02	1.67
Np-239	9.62	474	26,800	8,570	1,700	21	1,820
Pu-236	112	5,520	312,000	99,700	19,700	250	21,200
Pu-238	51.9	2,560	145,000	46,200	9,150	340	9,830
Pu-239	58	2,860	162,000	51,700	10,200	130	11,000
Pu-240	9,780	482,000	27,200,000	8,710,000	1,720,000	23,000	1,850,000
Pu-241	10,600	522,000	29,500,000	9,440,000	1,870,000	23,000	2,010,000
Am-241	51.7	2,550	144,000	46,000	9,110	450	9,800
Am-242	0.34	16.7	947	303	59.9	0.74	64.4
Am-242m	0.341	16.8	950	304	60.1	0.74	64.6
Am-243	9.62	474	26,800	8,570	1,700	21	1,820
Cm-242	490	24,100	1,360,000	436,000	86,300	1,100	92,800
Cm-243	4.9	241	13,600	4,360	863	11	928
Cm-244	2,750	135,000	7,660,000	2,450,000	485,000	18,000	521,000
Cm-246	0.215	10.6	599	191	37.9	150	40.7
Totals	231,000	11,400,000	644,000,000	206,000,000	40,700,000	520,000	43,800,000

a. Refer to Table C-8 for the names of the elements.

Table C-8. Chemical symbols used in Table C-7 and the corresponding element names.

H-3	=	tritium
Kr	=	krypton
Sr	=	strontium
Y	=	yttrium
Zr	=	zirconium
Nb	=	niobium
Tc	=	technetium
Rh	=	rhodium
Ru	=	ruthenium
Ag	=	silver
Cd	=	cadmium
Sn	=	tin
Sb	=	antimony
Te	=	tellurium
Cs	=	cesium
Ba	=	barium
Ce	=	cerium
Pr	=	praseodymium
Pm	=	promethium
Sm	=	samarium
Eu	=	europium
Tl	=	thallium
Pb	=	lead
Bi	=	bismuth
Po	=	polonium
At	=	astatine
Rn	=	radon
Fr	=	francium
Ra	=	radium
Ac	=	actinium
Th	=	thorium
Pa	=	protactinium
U	=	uranium
Np	=	neptunium
Pu	=	plutonium
Am	=	americium
Cm	=	curium

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